DEVELOPMENTAL HISTORY OF THE GAS TURBINE MODULAR HIGH TEMPERATURE REACTOR

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Abstract

The development of the high temperature gas cooled reactor (HTGR) as an environmentally agreeable and efficient power source to support the generation of electricity and achieve a broad range of high temperature industrial applications has been an evolutionary process spanning over four decades. This process has included on-going major development in both the HTGR as a nuclear energy source and associated power conversion systems from the steam cycle to the gas turbine.

This paper follows the development process progressively through individual plant designs from early research of the 1950s to the present focus on the gas turbine modular HTGR.

1. INTRODUCTION

This report traces the world-wide development of the HTGR through individual plant designs and selected facilities from initial research to the present focus on the modular HTGR coupled to the gas turbine. This developmental path has followed an evolutionary process spanning the last half of the twentieth century and, with some discretionary latitude, can be divided into the following five general areas:

- Gas Cooled Reactor (GCR) Development and Early HTGR Research
- HTGR Prototype Plants
- HTGR Demonstration Plants and Large Plant Designs
- Modular HTGR Steam Cycle Plant Development
- Modular HTGR Gas Turbine Plant Development

This maturing process has often been quite successful, but also tempered with developmental set-backs. However, one constant throughout this evolution has been the continuing pursuit for HTGR improvement as a safer, more efficient, environmentally acceptable and economically realistic source of nuclear energy for society.

A graphical presentation and the design characteristics of selected HTGRs are provided as Appendix A and B, respectively, at the end of this report.

2. GAS COOLED REACTOR (GCR) DEVELOPMENT AND EARLY RESEARCH

Gas-cooled reactor history effectively begins with the startup in November 1943 of the graphite-moderated, air-cooled, 3.5-MW, X-10 reactor in Oak Ridge, Tennessee. The X-10 was the pilot plant for the water-cooled, plutonium production reactors at Hanford, Washington, and used open-circuit cooling (that is, the air coolant was drawn from and exhausted to the atmosphere). Commercial gas-cooled nuclear power began in 1953 when the United Kingdom decided to combine plutonium production with electric power generation, and work was started on the four-unit power station at Calder Hall.

These first power reactors were graphite moderated with natural-uranium metal rods and cooled by forced circulation of carbon dioxide at a pressure of 0.8 MPa and at an outlet temperature of 335°C (635°F). The Calder Hall reactors became operational in 1956, and continue to produce a combined electrical power of 270 MW. The U.K.'s extensive commitment to GCR technology has included construction of 26 Magnox reactors (20 remain in operation) and 14 advanced gas-cooled reactors (AGRs), which deliver steam at the modern conditions of 538°C (1000°F) and 16.5 MPa (2400 psi).

France's early interest in the GCR aided development in the U.K. In 1951, the 2-MW research reactor at Saclay, which began operating with nitrogen coolant and later switched to carbon dioxide, was the first gas-cooled reactor to use closed circuit, pressurized cooling. These experiments, coupled with the experience of the air-cooled, open-circuit, G1 plutonium production reactor, formed the basis for France's GCR program. Although similar to the U.K's program in coolant, moderator, and fuel, the French program introduced the use of the prestressed concrete reactor vessel (PCRV), which the United Kingdom adopted for its later Magnox reactors and all of its AGRS.

The first nuclear power reactor in Japan, which started commercial operation in July 1966, was the carbon dioxide-cooled, 166 MW(e) Tokai station located 80 miles northeast of Tokyo. The plant design generally followed the design of the U.K's Magnox reactors; however, because of population concerns, its containment design provided a partial third barrier for a postulated release of coolant by sealing after pressure decay. In 1969, the Japan Atomic Energy Research Institute (JAERI) initiated studies on the very high temperature gas-cooled reactor (VHTR) in recognition that a nuclear process heat source of 900°C or higher would find use in coal gasification and hydrogen and methanol production [1].

Development of the HTGR began in the 1950s to improve upon GCR performance. HTGRs utilize ceramic fuel particles surrounded by coatings and dispersed in a graphite matrix, along with a graphite moderator. Either prismatic type graphite moderator blocks (block type reactor) or spherical fuel elements (pebble bed type reactor) are employed. Helium is used as the coolant to permit an increase in the operating temperature, and flows through coolant holes in the block type elements, or through the interstices present in the pebble bed core. HTGRs can operate at very high core outlet coolant temperatures because of the use of an all ceramic core [3].

3. HTGR PROTOTYPE PLANTS

The initial HTGRs included the Dragon reactor experiment, the Arbeitsgemeinshaft Versuchsreaktor (AVR) and Peach Bottom (No. 1). Common among these plants were the utilization of steel vessels to contain the primary system and the attainment of high core outlet temperatures culminating in the AVR achieving extended operation at 950°C.

3.1 Dragon Reactor Experiment

The Dragon reactor in the U.K. was the first HTGR prototype. It began in 1959 as an international project of the European Organization for Economic Cooperation and Development [4]. The initial objective of this project was to demonstrate the feasibility of the HTGR and to launch the development of a technology which had already begun at a low level in various national laboratories. Simultaneously with the initiation of the Dragon project, interest in Germany and the U.S. led to the 15 MW(e) AVR at Julich and the 40 MW(e) HTGR at Peach Bottom, respectively.

The Dragon Reactor Experiment first operated at power in July 1965 and reached its full-power operation of 20 MW(t) by April 1966. The reactor had a steel pressure vessel, graphite fuel elements with high-enriched uranium-thorium carbide coated fuel particles, and a helium coolant. Exit helium at 750°C (1382°F) was circulated through primary heat exchangers and returned at 350°C (electric power generation was not a feature of this prototype.) The reactor operated for long periods at full power and demonstrated the successful operation of many components, e.g., the helium circuit purification system, the control rod drives, the fuel handling equipment, the gas-bearing circulators, and the reactor fuel. As a test bed, it provided much information on fuel and material irradiation tests and component tests under high-purity helium conditions. The reactor operated until March 1976, at which time the project was terminated [5].

3.2 The AVR

In August 1959, the order was placed with the German partnership of BBC/Krupp for the construction of an experimental nuclear power plant with a high temperature reactor of 46 MW(t)/15 MW(e). Construction work on the plant at the KFA Nuclear Research Center in Juelich began in 1961. Initial reactor criticality occurred in August 1966, and on 17 December 1967 electricity was supplied to the public supply grid for the first time. In February 1974, the mean gas outlet temperature, which had initially been set at 850°C, was increased to 950°C. Operation of the AVR continued until December 1988. In completing 21 years of service, the plant had accumulated more than 122,000 hours of operation with a 66.4% overall availability and had generated 1.67 billion kwhr of electricity [3].

The design of the AVR included a steel containment vessel and used particle fueled, graphite spheres 6 cm in diameter that traveled downward through the core. Although the AVR initially included a core outlet temperature of 850°C, this was subsequently raised to 950°C without decrease in plant performance. The AVR was the main fuel development tool for the pebble bed concept, and it and supplementary laboratory fuel testing became the major support of Germany's position that an LWR-type containment barrier was not needed for future HTGRs [6].

The core of the reactor consisted of a pebble bed containing 100,000 fuel spheres in a graphite reflector pot. The steam generator was located above the core and shielded against radiation from the core by a 50 cm thick top reflector made of graphite and two additional 50 cm thick layers of carbon bricks. The coolant was helium, pressurized to 10 bar and circulated by two blowers located in the lower section of the reactor vessel. The helium was heated in the core from 270°C to 950°C and then flowed through the steam generator, where it transferred the energy to the water/steam circuit. The generated steam had a pressure of 73 bar and a temperature of 505°C. The core, steam generator and blowers were surrounded by two concentrically arranged reactor vessels. The interspace between the vessels was filled with helium at a pressure slightly above the coolant gas pressure.

In addition to continuous component testing and the collection of operating experience and results, the plant was also used during its operating period for the implementation of several experiments. In particular, these included the testing of various types of fuel elements, investigations of fission product behavior in the circuit, experiments on the chemistry of coolant gas impurities and the performance of several tests related to HTGR safety.

During the last few years, the operating mode of the AVR was increasingly oriented to performing test programs related to HTGR performance and safety. During 1988, operations were centered almost exclusively on tests related to the safety of HTGRs and, in particular, to the performance of simulated loss of coolant accidents [3].

3.3 Peach Bottom (No. 1)

Peach Bottom Unit 1 was the first prototype HTGR in the U.S. The 40 MW(e) plant, owned and operated by the Philadelphia Electric Company, was built as part of the US Atomic Energy Commission (USAEC) Power Reactor Demonstration Program and the High Temperature Reactor Development Associates (HTRDA) Program to demonstrate the feasibility of a high performance, helium cooled, nuclear power plant.

The reactor achieved initial criticality on March 3, 1966, and the plant went into commercial operation in June 1967. It operated successfully with a gross capacity factor of 74% and an overall availability of 88% (exclusive of planned shutdowns for R&D programs) from June 1967 until October 31, 1974, when it was shut down for decommissioning. Decommissioning was a planned economic decision, based on the additional costs associated with satisfying revised USAEC safety requirements. Accumulated operation totaled 1,349 effective full-power days, for a total of 1,385,919 gross electrical megawatt hours generated.

Peach Bottom provided a valuable demonstration of the high-temperature reactor concept by confirming the core physics calculations, verifying the design analysis methods, and providing a data base for further design activities in the following areas: reactor core; mechanisms in high-temperature helium; helium purification systems; and steam generator tube materials.

Two reactor cores were utilized in the operation of Peach Bottom. The fuel particles in Core 1 were coated with a single layer of anisotropic carbon. Fast-neutron-induced dimensional changes resulted in fracture and distortion of the coatings, which eventually resulted in 90 cracked fuel element sleeves out of 804. Plant operation, however, was not impaired; reactor operation continued, and primary circuit activity reached a maximum of 270 Ci, well below the design activity level of 4,225 Ci. Core 1 accumulated 452 equivalent full power days before it was replaced with Core 2. Core 2, which provided buffer isotropic pyrolytic carbon (BISO) coatings on the fuel particles, operated with no fuel failures for its full design lifetime of 897 equivalent full power days. The primary circuit activity averaged only 0.5 Ci during this period.

Throughout the operation of Peach Bottom, excellent agreement was found between the predicted and the actual plant design characteristics, verifying the methods used and providing a reference data base for application to larger HTGR plants. The overall plant control system functioned exceptionally well, and the plant was operated in a load-following manner during the majority of its lifetime, demonstrating the ability of the HTGR to function in this manner [5].

4. HTGR DEMONSTRATION PLANTS AND LARGE PLANT DESIGNS

Early international development of the HTGR focused on two basic core designs recognized specifically by their fuel element structure. The German core design has generally followed a core design incorporating spherical fuel elements (pebble bed), whereas, beginning with Fort St. Vrain (FSV), the U.S. core design included ceramic coated fuel particles imbedded within rods placed in large hexagonal shaped graphite elements. The other major HTGR designer during this period was Russia with their VG series of plant designs which incorporated the pebble bed core.

The HTGR plants that followed the successful AVR and Peach Bottom included construction of FSV and the Thorium High Temperature Reactor (THTR-300). A major shift occurred with these plants including primary systems enclosed within PCRVs rather then steel vessels, and accompanied by significant increases in plant power level.

These two features were to be dominant throughout HTGR development of the 1970s and early 1980s. Although the general focus of the HTGR plant designers was on the development of large steam cycle units, including the first significant design for a closed cycle gas turbine plant, these designs did not materialize into the commissioning of a nuclear plant. This was primarily attributed to the nuclear industry being in a general state of decline in those countries with national HTGR development programs.

Performance of the THTR-300 and FSV in validating reactor safety characteristics and the TRISO coated fuel particle was very good. However, FSV was beset with inconsistent operation and a corresponding low capacity factor throughout its life. The THTR-300 was also plagued with problems primarily associated with changing regulatory requirements that contributed to delays that eventually ended up with a construction period of ~ 14 years. Corresponding financial concerns by the operating utilities led to the premature shutdown of both these plants in the 1988-1989 time frame.

4.1 Fort St. Vrain (FSV)

The FSV HTGR was operated by Public Service Company of Colorado as part of the United States Atomic Energy Commission's Advanced Reactor Demonstration Program. This plant was designed with several advanced features including; a.) A PCRV containing the entire primary coolant system, b.) A core of hexagonal, graphite block fuel elements and reflectors with fuel in the form of ceramic coated (TRISO) particles, c.) Once-through steam generator modules producing 538°C superheated main and reheat steam, and d.) Steam-turbine-driven axial helium circulators. At its rated capacity, the FSV plant generated 842 MW(t) to achieve a net output of 330 MW(e).

The steam cycle was essentially conventional, utilizing a standard reheat turbine, except that the steam flow was from the high-pressure turbine exhaust to the helium circulator turbines before being reheated and returned to the intermediate pressure turbine. The steam conditions were comparable to those of modern fossil units.

Initial electric power generation was achieved at FSV in December 1976, and 70% power was reached in November 1977. Full-power operation was achieved in November 1981 [5]. Although ~5.5 billion kwhr of electricity was generated at FSV, the plant operated at low availability primarily because of excessive downtime due to problems with the water-lubricated bearings of the helium circulators.

In spite of this low availability, the plant was a valuable technology test-bed, successfully demonstrating the performance of several major systems, including the reactor core with TRISO coated fuel particles in hexagonal graphite blocks, reactor internals, steam generators, fuel handling and helium purification [1].

4.2 Thorium High Temperature Reactor (THTR-300)

The THTR-300 nuclear power plant included a steam cycle for the generation of electric power with a net output of 296 MW. The heat generated in the reactor core of the helium cooled, graphite moderated high temperature reactor was transported via the helium gas coolant circuit (primary system) to the steam generators where it was transferred to the steam-feedwater circuit (secondary system) and then transported to the turbine generator. The secondary system was cooled by means of a 180 m high natural draught dry cooling tower [3].

The THTR-300 power plant was sponsored by the Federal Republic of Germany and Nordrhein Westfalen (NRW). Construction of this 300 MWe plant began in 1971, but primarily due to increasing licensing requirements, the plant was not completed until 1984. This pebble bed reactor plant was connected to the electrical grid of the utility Hochtemperatur-Kernkraftwerk GmbH (HKG) in November 1985. In August 1989, the decision was made for the permanent shutdown of the THTR-300. This action was not due to technical difficulties associated with the plant, but was the result of an application by HKG for early decommissioning based on a projected short fall in funding and contractual changes in the allocation of decommissioning costs between the FRG, NRW and HKG that would take effect upon the termination of the demonstration phase in 1991. Operation of the THTR-300 was successful in validating the safety characteristics and control response of the pebble bed reactor, primary system thermodynamics and the good fission product retention of the fuel elements [2]. The plant operated over 16,000 hrs. and generated 2.891 billion kwhr of electricity [20].

4.3 Large HTGR Steam Cycle Plant Designs [2]

The primary focus of international HTGR designers in the 1970s and early 1980s was associated with the development of large HTGR units. Continued interest in development of larger steam cycle HTGR plants included the German HTR-500, the Russian VG-400 and the United States' HTGR-Steam Cycle (SC) plants.

In the U.S., the focus in the early 1970s was on HTGR-Steam Cycle designs of 770 to 1,160 MWe. Contracts to General Atomics from U.S. utilities included 10 plants that did not materialize due primarily to the 1973 oil crisis and corresponding economic setback and collapse of the U.S. nuclear power market of 1975. These plants incorporated cores of hexagonal graphite blocks with TRISO coated fuel particles similar to FSV.

The HTR-500 made considerable use of the technology development for the THTR-300, with simplifications and optimizations based on practical experiences with the THTR-300 [3]. This plant featured a simple design with the primary system components located within a single cavity PCRV, and included a pebble bed reactor power level of 1,390 MWt to produce 550 MWe of electricity.

The Russian HTGR development program included the VG series of plants primarily developed at OKBM in Nizhny Novgorod. Of these, the VG-400 design incorporated a pebble bed reactor with a final power level of 1,060 MWt for co-generation applications of electricity generation and process heat production for steam reforming of methane. The reactor was designed for a core helium outlet temperature of 950°C. During the preliminary phases of the design development both the pebble bed and prismatic fuel block variants of the core were analyzed. As a result of the design and engineering analysis, the pebble bed core was chosen for further development due to considerations of simplified fuel element manufacturing technology and possibility of full scale testing in experimental reactors, utilization of a simplified core refueling mechanism and the possibility of core refueling during on-load reactor operation. Also considered was a gas turbine cycle/VG-400 reactor plant.

4.4 Early Gas Turbine HTGR Assessment

A promising approach for making good use of the high temperature capability of the HTGR is to use the primary helium coolant to drive a gas turbine in a direct closed cycle arrangement. In the 1970s, this was extensively studied in the U.S., Germany, the U.K. and France. At that time, the concept was based on enclosing a large (2,000 to 3,000 MWt) reactor core and the gas turbine power conversion system within a prestressed concrete reactor vessel [19]. One of the early gas turbine designs was the HTGR-GT by General Atomics. This plant was projected to have the potential for high plant efficiencies (40%) under dry cooling conditions and achievement of even higher efficiencies (50%) when bottoming cycles were incorporated. An assessment of this plant was subsequently conducted in ~ 1980 with the focal point for the assessment being the potential commercial HTGR-GT plant having the following characteristics:

• Size	2000 MW(t)/800 MW(e)
• Core outlet temperature	850°C
• Number of heat transport turbine loop	2

The principal findings of the assessment (in the early1980s) were as follows: (1) the HTGR-GT is feasible, but with significantly greater development risk than that associated with the HTGR-SC; (2) at the level of performance corresponding to the reference design, no incremental economic incentive can be identified for the HTGR-GT to offset the increased development costs and risk relative to those for the HTGR-SC (this was true over the range of cooling options investigated); (3) the relative economics of the HTGR-GT and HTGR-SC are not significantly affected by dry cooling considerations; and (4) although the reduced complexity of the cycle may ultimately result in a reliability advantage for the HTGR-GT, the value of that potential advantage could not be quantified.

Although these (early 1980s) findings did not provide the basis for the HTGR-GT as the preferred lead commercial plant (for General Atomics), the HTGR-GT continued to engender interest from participating utilities. This interest stemmed from the potential of the HTGR-GT for improved efficiencies at high core outlet temperatures, the basic simplicity of the gas turbine cycle (low maintenance, high capacity factors), low water use requirements and attractive co-generation characteristics [5].

In retrospect, this plant, although significantly different in general configuration compared to the current modular HTGR/gas turbine designs, the basic cycle remain unaltered throughout these past twenty years. Figures 2 and 3 provide the HTGR-GT configuration and basic cycle diagram, respectively.

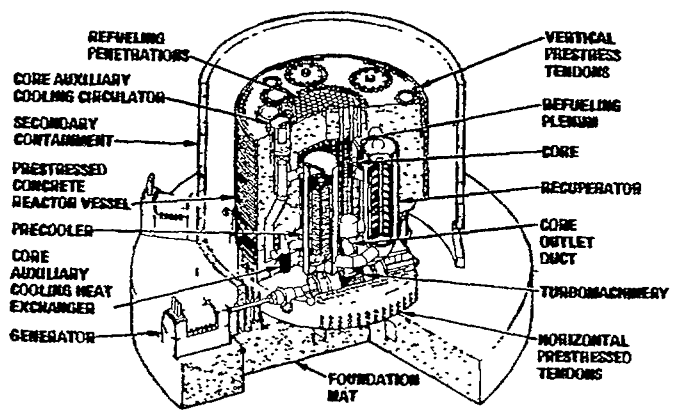


FIG. 2: HTGR-GT system [5]

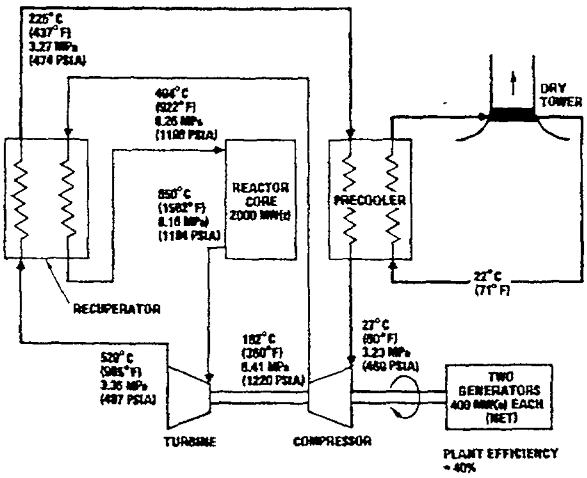


FIG. 3: HTGR-GT cycle diagram [5]

5. MODULAR HTGR STEAM CYCLE PLANT DEVELOPMENT

The overall good safety characteristics of all HTGRs are due to: the high heat capacity of the graphite core; the high temperature capability of the core components; the chemical stability and inertness of the fuel, coolant, and moderator; the high retention of fission products by the fuel coatings, the single phase characteristics of helium coolant; and the inherent negative temperature coefficient of reactivity of the core. Figure 4 provides a graphic overview of maximum accident core temperatures for U.S. HTGR designs.

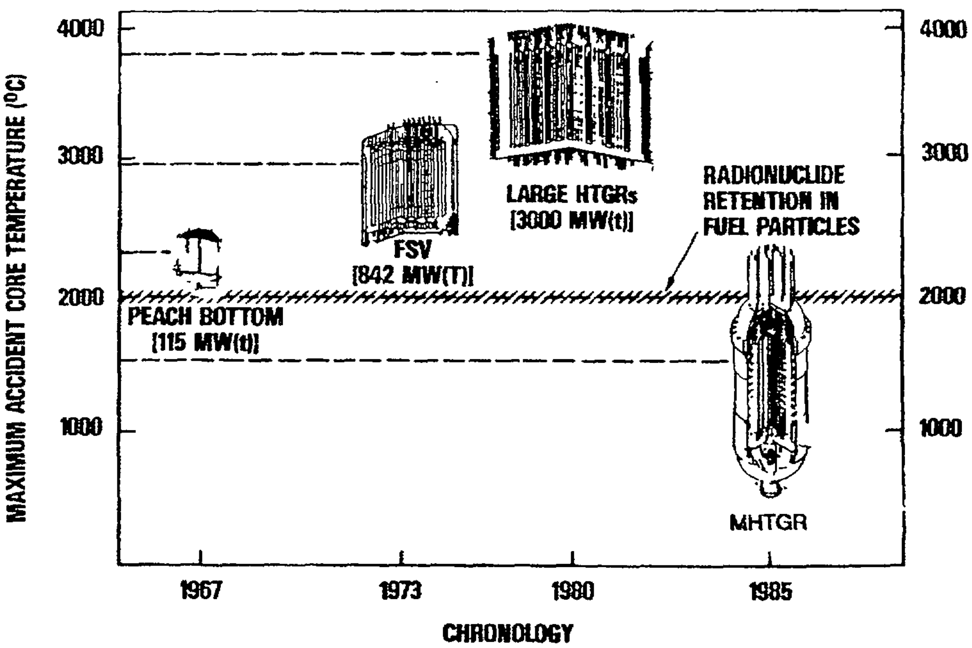


FIG. 4: Maximum accident core temperature depicted chronologically for U.S. HTGR designs [1]

The modular HTGR adds the unique characteristic of being able to cool the reactor entirely by passive heat transfer mechanisms following postulated accidents without exceeding the failure temperature of the coated particles. This characteristic has been achieved by deliberately decreasing the core power level and configuring the reactor so that natural heat removal processes can limit fuel temperatures to levels at which the release of fission products from the reactor system to the environment is insignificant for postulated accidents. Even for extreme accidents having very low probabilities of occurrence, the cumulative fission product release at the site boundary is estimated to be below those acceptable under defined protective action guidelines [2].

5.1 HTR-MODULE [3]

The 80 MW(e) HTR module (HTR-MODULE) concept, developed by Siemens/Interatom, was the first small, modular type HTGR concept to be proposed. Although initially developed in the early 1980s for industrial process heat applications, the passive safety features of the side-by-side concept, coupled with the other attractive characteristics of the modular concept, soon led to the proposed electricity generation application. Work on a generic, site independent safety assessment of the HTR module was initiated with the filing, in 1987, of the safety analysis report in the State of Lower Saxony. The HTR module safety concept is characterized by comprehensive protection of the environment by passive system characteristics even under extreme, hypothetical accident conditions.

The safety features of the HTR module were based on the design condition that, even in the case of failure of all active cooling systems and complete loss of coolant, the fuel element temperatures would remain within limits such that there is virtually no release of radioactive fission products from the fuel elements. Such a condition guaranteed that the modular HTR power plant would not cause any hazard to the environment either during normal operation or in the case of an accident.

A small core diameter (~ 3 m) stems from the requirement for reactor shutdown from all operating conditions using only free falling control rods in reflector borings. The requirement to keep the maximum fuel element temperature for all possible accidents inherently below 1600°C, a temperature at which all radioactive fission products are contained within the fuel elements, leads directly to a mean power density of 3 MW/m^3. In order to gain as much power as possible from the core, the core height was chosen as large as possible.

5.2 HTR-100

With the objective that nuclear power plants utilizing small HTGRs can provide economic, environmentally favorable and reliable electricity and heat for community and industrial purposes, Brown, Boveri und Cie and Hochtemperatur-Reacktorbau initiated the design of the HTR-100 pebble bed plant. This design featured a 285 MWt HTGR with a net electrical output of 100 MW on the basis of the AVR concept and utilized the advanced technologies of THTR-300 components and systems. The primary system included the reactor, steam generator and helium circulator in a single, vertical steel pressure vessel. The reactor core included 317,500 spherical elements of TRISO type particles and a power density of 4.2 MW/m^3. In the equilibrium cycle, 55% of the spherical elements were fuel, with the remainder being graphite. The basic design for this plant incorporated two HTR-100 units with overall capability of producing 170 to 500 t/hr of industrial steam at 270°C and 16 bar, with 100 to 175 MW(e) gross output [9].

5.3 VGM

Analysis of the heat energy market in Russia revealed a need for the development of small nuclear power energy sources whose power utilization factor and high degree of safety could meet the requirements of high availability and capacity factors and potential close-in siting required by industrial plants. The VGM modular HTGR, with a power output of 200 MW(t) and a design configuration very similar to the HTR-MODULE which was being developed in Germany, was selected as a low power pilot unit after considering several different designs. The power required by large industrial complexes would be reached by the use of several such modules, permitting also the necessary reserves for both high availability and high capacity factors.

5.4 MHTGR

In 1983, the U.S. organization representing utility/user interests in the HTGR program, the Gas Cooled Reactor Associates, conducted a survey to determine the utility nuclear generation preference for the future. This survey resulted in a strong interest for smaller generation increments. This was an important input leading to the evaluation and subsequent selection of the modular HTGR in 1985 [1]. Following a detailed evaluation in the spring and summer of 1985, a side-by-side concept similar in configuration to a German module design was chosen as the reference concept for further design and development by the U.S. program. The basic module was designed to deliver superheated steam at 17.3 MPa and 538°C. An initial module power level of 250 MWt was selected, but subsequent detailed safety analyses showed that this power level could be increased with the hexagonal graphite block core design without compromising margins. Adopting an annular core allowed the power level to be increased initially to 350 MWt. Other reactor design changes and analysis refinements subsequently allowed the power to be increased to 450 MWt, while maintaining adequate margins to component and safety limits [8].

Central to the Modular HTGR (MHTGR) passive safety approach was the annular reactor core of prismatic fuel elements within a steel reactor vessel. A low-enriched uranium, once-through fuel cycle was used. For a standard steam cycle MHTGR plant, the steam output from each of the four modules was connected to an individual turbine generator. The four module plant consists of two separate areas, the nuclear island and the energy conversion area.

The most fundamental characteristic of the MHTGR that separated it from previous reactor designs was the unique safety philosophy embodied in the design [7]. First, the philosophy requires that control of radionuclides be accomplished with minimal reliance on active systems or operator actions; the approach to safety is to rely primarily on the natural processes of thermal radiation, conduction, and convection and on the integrity of the passive design features. Arguments need not center on an assessment of the reliability of pumps, valves, and their associated services or on the probability of an operator taking various actions, given the associated uncertainties involved in such assessments.

Second, the philosophy requires control of releases by the retention of radionuclides primarily within the coated fuel particle rather than reliance on secondary barriers (such as the primary coolant boundary or the reactor building). Thus, ensuring that the safety criteria are met is the same as ensuring that the retention capability of the coated fuel particles is not compromised.

The assessment of the capability of the MHTGR to control accidental radioactivity releases shows that the doses are a small fraction of the U.S.10CFR100 requirements even for the bounding analyses which consider only the systems, structures and components that require neither operator action nor other than battery power. In fact, the exposures are so low that the protective action guidelines would require no evacuation or sheltering plans for the public as specified in the utility/user requirements. The evaluation confirms that accident dose criteria can be met with a containment system that places primary emphasis on fission product retention within the fuel barriers.

Consistent between the modular HTGR designs of the HTR-MODULE, MHTGR and the VGM are side-by-side steel vessels housing the reactor and the helium circulator and steam generator in a common configuration. Figures 5 and 6 depict the U.S. MHTGR and German HTR-MODULE, respectively. The VGM is similar with the inclusion of another vessel housing an auxiliary cooling system.

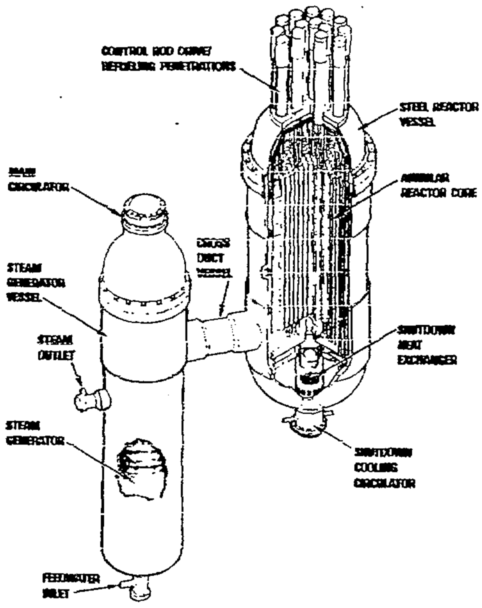


FIG. 5: MHTGR arrangement [1]

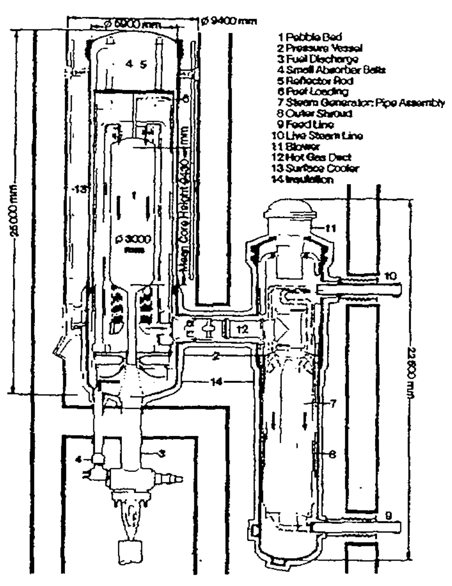


FIG. 6: HTR-MODULE arrangement [3]

It was the development of the above mentioned modular HTGR steam plants that provided the key emphasis to initiation of the HTGR coupled to a gas turbine power conversion system. With few exceptions, it is the German HTR-MODUL and the HTR-100 reactor designs that are being utilized by ESKOM as the base for the PBMR. The MHTGR, with its annular core arrangement within a steel vessel, forms the basis for the GT-MHR reactor design [2].

Also, it was during the 1980s and early 1990s that a major focus of the international HTGR community was directed towards investigation of the industrial heat applications and co-generation capabilities provided by the modular HTGR. Investigation of the capabilities of this nuclear energy source to support high temperature process heat applications as well as the need to evaluate the safety and technological features of the HTGR were contributing factors in the decisions by JAERI and China's Institute of Nuclear Energy Technology (INET) to construct the High Temperature Engineering Test Reactor (HTTR) and High Temperature Gas Cooled Reactor Test Module (HTR-10), respectively.

6. MODULAR HTGR GAS TURBINE PLANT DEVELOPMENT

It has long been recognized that substantial gains in the generation of electricity from nuclear fission can be achieved through the direct coupling of a gas turbine to a HTGR. This advanced nuclear power plant is unique in its use of the Brayton cycle to obtain a net electrical efficiency approaching 50% [10]. This plant provides a promising alternative for the utilization of nuclear energy to produce electricity. Although evaluation of this concept was initiated over twenty years ago, it was terminated due to the technical risks primarily in the component areas of magnetic bearing, compact plate-fin heat exchanger and turbo machinery development. Subsequent technological advancements in the design and operation of these components, coupled with the international capability for their fabrication and testing has resulted in renewed interest in this HTGR concept [11].

6.1 Development of the Gas Turbine Plant

In order to be competitive, the thermal efficiency of nuclear power had to be markedly improved to compete with modern, high efficiency fossil plants. HTGR technology has always held the promise for electricity generation at high thermal efficiency by means of a direct Brayton cycle and fortuitously, technological developments during the past decade provided the key elements to realize this promise. These key elements are as follows:

- The HTGR reactor size had been reduced in developing the passively safe module design. At the same time, the size of industrial gas turbines had increased. The technology was now available for a single turbo-machine to accommodate the heat energy from a single HTGR module.

- Highly effective compact recuperators had been developed. Recuperator size and capital equipment cost are key economic considerations. Highly effective plate-fin recuperators are much smaller than equivalent tube and shell heat exchangers, provide for substantially less complexity and capital cost, and are a key requirement for achieving high plant efficiency.

- The technology for large magnetic bearings had been developed. The use of oil lubricated bearings for the turbo-machine with the reactor coolant directly driving the turbine was problematic with regard to the potential coolant contamination by the oil. The availability of magnetic bearings eliminates this potential problem [12].

A major requirement was for the plant to become substantially simplified in order to provide a significant reduction in the capital expenditure for new capacity additions. Figures 7 and 8 provide a comparison of nuclear power plant efficiencies and a graphic simplification that can be achieved in moving from the HTGR steam cycle to the basic direct gas turbine cycle, respectively.

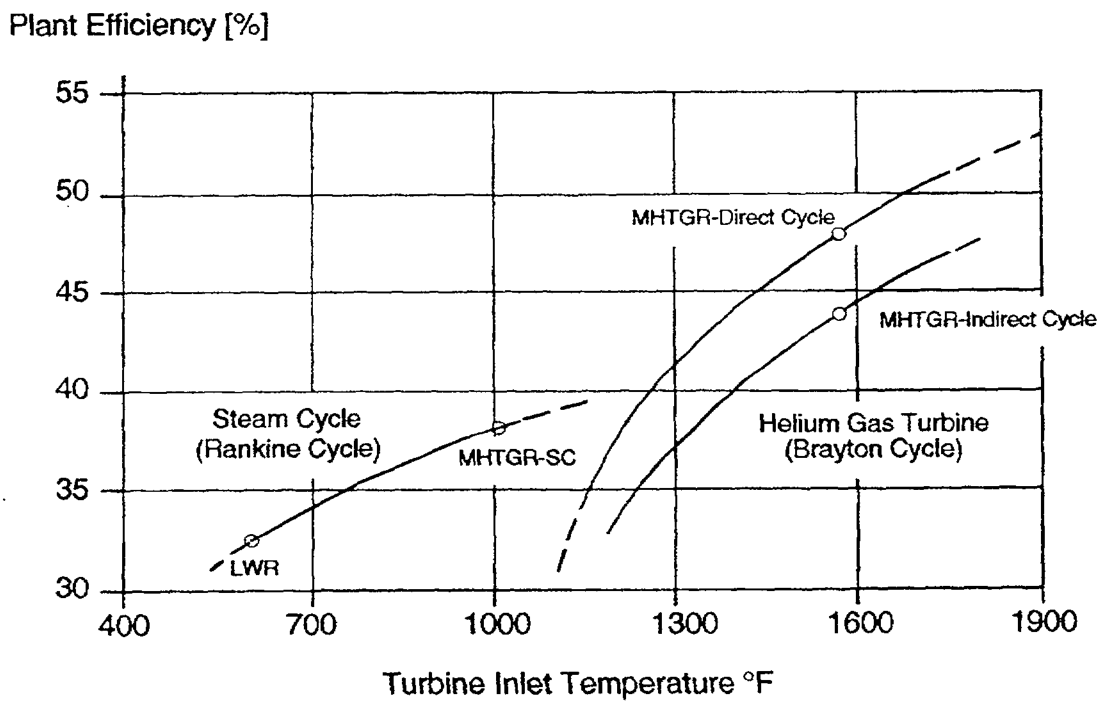


FIG. 7: Plant efficiency comparison [1]

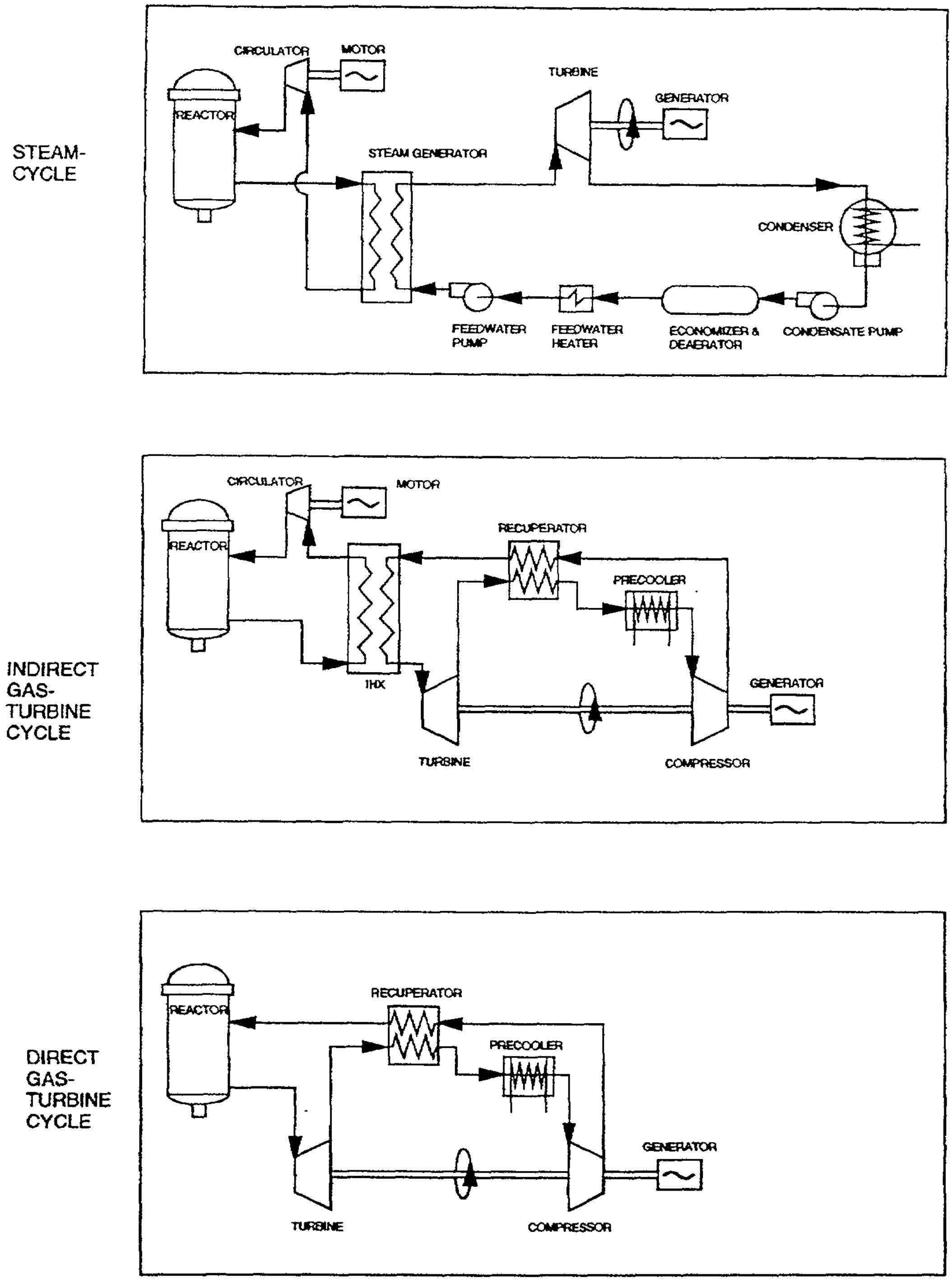


FIG. 8: System simplification, steam cycle to the closed cycle gas turbine plant [1]

The possibilities presented by the gas turbine modular HTGR for substantial improvement in nuclear power plant efficiency coupled with the potential for significant gains in lowering capital and operating costs due to plant simplification have brought about an increasing interest by international research organizations and plant developers.

Overviews of the most prominent among the designs currently being investigated are provided in the following sub-sections.

6.1 Pebble Bed Modular Reactor (PBMR)

The PBMR was first identified by ESKOM in 1993 as an option for expansion of their electrical generation capacity. Subsequently, ESKOM contracted with Integrators of System Technology to perform a technical and economic study of the feasibility of the PBMR for the generation of electricity. This study, which was completed in early 1997, supported the continued development of the PBMR. Reactor development follows the HTR-MODULE pebble bed which was previously licensed in Germany for commercial operation.

A series of internal reviews by ESKOM and subsequent independent reviews of this feasibility study by international entities have generally acknowledged that the PBMR is technically and economically capable of meeting the requirements originally set by ESKOM for commercialization. These requirements included: a.) New generation capacity capable of being located where the load growth is taking place in South Africa (coastal regions); b.) Small, modular increments of electrical generation capacity corresponding to system growth needs; c.) Reduced exposure to negative environmental issues such as carbon dioxide emissions and capable of providing a strategy for economic mitigation of greenhouse gas reductions; d.) Generating plants placed where there would be a limited need for extensive transmission system additions, and; e.) Cost of capital and plant operation to be within those costs presently being achieved at ESKOM's largest coal fired stations (with targets of capital cost less than US$ 1,000/installed kw and overall generation, transmission and distribution costs of utility operations within 2.0 cents/kwhr [13]. ESKOM, in conjunction with British Nuclear Fuel, the U.S. utility PECO Energy and the Industrial Development Corporation of South Africa, is currently in the process of performing the final evaluation for the introduction of this advanced nuclear power reactor coupled to a closed cycle gas turbine as additional generation capacity on their electric grid and for commercialization in the international market place [2].

6.2 Gas Turbine-Modular Helium Reactor (GT-MHR)

Renewed interest in the nuclear powered closed cycle gas turbine system within the U.S. resulted in the present GT-MHR developmental program beginning in early 1993. Subsequent discussions with organizations of the Russian Federation resulted in GA and MINATOM entering into a Memorandum of Understanding to cooperate on the development of the GT-MHR with the goal, following design and development, to construct, test and operate a prototype in Russia [14]. FRAMATOME joined this program in 1996 with Fuji Electric becoming a participant and sponsor in 1997.

The conceptual design is now completed on this 600MWt/293Mwe plant which is currently under development for the destruction of weapons plutonium, but with the longer term goal of commercial development. The next stage of development is in preliminary design of the plant to begin in 1999. Most of the design work on the GT-

MHR is being performed within the nuclear organizations of the Russian Federation with financial and management/technical support from all members of the consortium.

A recent significant development in the advancement of the GT-MHR is the authorization of financial support by the U.S. government on a matching resource basis with MINATOM for the destruction of weapons plutonium. Although the GT-MHR is initially to utilize a plutonium fuel cycle which has the capability of achieving a burn-up approaching 95%, the versatility and flexibility of this core will allow for the application of a wide range of diverse fuel cycles. Fuel derived from uranium, thorium and a variety of plutonium grades is under consideration for long term applications in the GT-MHR [13].

6.3 Other Gas Turbine and Modular HTGR Co-generator Designs [13]

6.3.1 ECN's ACACIA Plant

ECN Nuclear Research is developing a conceptual design of an HTGR for the combined generation of heat and power for industry within the Netherlands as well as for possible export. The ACACIA plant utilizes a 40MWth pebble bed HTGR to produce 14MW of electricity and 17 tonnes of 10 bar, 220°C. steam per hour [15]. The electric generation system utilizes a basic closed cycle gas turbine which receives helium from the HTGR at 800°C and 2.3MPa. After the recuperator, a secondary helium loop removes heat from the primary system via an intermediate heat exchanger (precooler) which then transfers energy to the steam/feedwater system for industrial use.

6.3.2. Japan's HTGR Gas Turbine Designs

A number of HTGR Gas Turbine plant designs are currently under development within Japan. These plants are primarily under development by JAERI within the framework of the Japanese HTGR-GT feasibility study program, and include gas turbine cycle units with reactors of 400MWt, 300MWt, and two variations with a power level of 600MWt. In order to provide an indication of the diversity of application under consideration in Japan, two of these designs are discussed as follows:

The 300 MWt Plant: The 300 MWt pebble bed annular reactor of this plant provides a core outlet temperature of 900°C. to the turbine of a single shaft machine. The design is coordinated by JAERI and includes developmental support from Japanese industry. The flexibility of this plant includes the co-generation application of electricity production and the capability to provide 283 tons of desalinated water per hour through the use of an additional heat exchanger between the recuperator and precooler. The net thermal efficiency for this plant is anticipated at 48.2% [16].

The GTHTR 300 Plant: This plant incorporates a 600MWt hexagonal fuel block core. The power conversion system includes a vertical heat exchanger vessel and a horizontal turbo-machine vessel to allow for bearing support and stable rotor operation. The cycle configuration has been simplified in comparison to the GT-MHR by elimination of a compressor unit and the corresponding intercooler. The overall net plant efficiency for this simplified unit is 45.4% [17].

6.3.3 MIT and INEEL's MPBR Plant

The development of the MPBR is being addressed based on preliminary research into future energy options by MIT student work beginning in 1998. It was concluded that this technology provided the best opportunity to satisfy the safety, economic, proliferation, and waste disposal concerns that face all nuclear generating technologies. The areas of research for this project are aimed at addressing some of these fundamental concerns to determine whether the small 110MWe modular gas-cooled pebble bed plant can become the next generation of nuclear technology for worldwide deployment [18].

MIT and INEEL have utilized the reference design from the ESKOM PBMR, but with a significantly different balance of plant. The HTGR is of pebble bed design with a power level of 250 MWt. Primary coolant helium from the reactor flows through an intermediate heat exchange providing a transfer of energy to the secondary coolant (air). The secondary loop consists of a high pressure turbine which drives three (high, medium and low pressure) compressors with two stages of intercooling. A second shaft incorporates the low pressure turbine and electric generator. High temperature coolant air leaves the IHX to drive the compressors via the high pressure turbine. After exiting this turbine, the coolant then enters the low pressure turbine to expand and drive the generator. From this turbine, the secondary coolant path includes the recuperator, precooler and compressors/intercoolers. The MPBR utilizes conventional oil bearings rather then magnetic bearings on its turbomachines.

6.3.4 INET's MHTGR-IGT

This design features an indirect gas turbine system coupled to a 200MWt pebble bed HTGR. Although the HTGR can provide heat at 950°C with the attributes of outstanding safety and gas turbine cycle efficiency in the range of 47%, the possible radioactivity deposition on the turbine blades and thus the increase in maintenance difficulties suggests that the indirect gas turbine cycle should be applied initially in the development process to help solve these problems.

In this design, the helium out of the intermediate heat exchanger (IHX) is extracted to a RPV cooling system. The gas flows through a small RPV recuperator and is cooled. It then is used to cool the RPV. The whole primary circuit is integrated in a single pressure vessel with the core inlet/outlet temperatures 550°C/900°C, which can supply heat of ~ 850°C on the secondary side. The heat source would be used to drive a nitrogen gas turbine cycle with a busbar electricity generation efficiency of about 48%. The 200MWt pebble bed reactor core is located at the lower position of reactor pressure vessel with a geometry similar to that of the Siemens 200MW HTR-MODULE. The straight tube IHX is located at the upper position of the RPV and connected with the core through a gas duct. Similar to the AVR, the control rod system is installed at the bottom of the RPV and all rods are inserted upward into the side reflector. The main helium blower and an auxiliary blower for shutdown cooling are located at the top of the RPV. A reactor vessel cooling system recuperator and cooler are installed respectively in annular regions .outside of the IHX and blowers [2].

7. CONCLUSIONS

The HTGR development process has been subjected to many significant changes from the initial Dragon plant to the present gas turbine designs. Major among these changes are the following:

- Dragon, Peach Bottom and the AVR were commissioned in the 1960s primarily to develop and demonstrate the feasibility of HTGR technology. The plants were generally quite successful in achieving the individual goals set for them. This was particularly evident with the AVR in demonstrating extended reactor operation at an average core outlet temperature of 950°C, and for validation of the UO_2 kernel TRISO coated fuel particle. This fuel represents the foundation for the safety and environmental aspects of the HTGR, and is now used exclusively in all modular HTGR designs.
- FSV and the THTR-300 were then commissioned to be the next milestones in HTGR development. The primary goal for these plants was to demonstrate the commercial capabilities of the HTGR as the fore-runners to achieving marketability of the large follow-on plant designs. These plants were not successful in accomplishing this goal. Commonalties shared by FSV and the THTR-300 were size (each ~ 300MWe) and the utilization of a PCRV rather then steel vessels to contain the primary coolant system. Of note is that these plants were valuable in demonstrating the attributes of the HTGR, including the TRISO coated fuel particle.
- The modular steam cycle HTGR became the next focus for plant designers. This was influenced by utilities and designers to evaluate small nuclear power sources that would have added safety characteristics including the potential for being located at industrial sites. This resulted in the adoption of the modular HTGR with its safety attribute of passive heat transfer resulting in plant simplification and associated economic advantages. Modular HTGR designs of this era reverted back to the steel vessel, generally in a side-by-side configuration, and the reactor took on an annular configuration. Although the generation of electricity was the preferred product for this plant, an ever increasing interest, primarily by many national research organizations, was the development of co-generation and industrial applications afforded by its capability to achieve high core outlet temperatures. .
- The major emphasis of HTGR development into the 1990s continued with the modular plant, but now with an annular reactor core and coupled to a gas turbine power conversion system rather then the steam cycle. These changes are anticipated to contributed substantially to the goals of reduced capital and operating costs by further plant simplification and a significant improvement in cycle efficiency over existing nuclear plant designs.

The gas turbine plant continues to be the primary focus of international HTGR development. However, of note is that the constituency of partners in this development has changed significantly in the past decade. This is particularly evident with the PBMR and GT-MHR where international partnerships now include major nuclear plant developers and utilities. The reasons for this are obvious; a design that shows promise for significant improvements in nuclear plant economics and attendant plant efficiency, with the long term prospect of achieving substantial marketability and associated financial reward. However, the task ahead for commercialization of this plant has to overcome significant hurdles including: 1.) A modular HTGR has never before been constructed, 2.) The size and environmental application of major components such as plate-fin recuperators and turbomachines fitted with magnetic bearings has yet to accomplished, and 3.) The licensing process has yet to be completed for a nuclear power plant of this configuration.

Yet, successful commercial deployment of this plant has the potential of significantly advancing nuclear power as a world-wide primary energy source for the future with corresponding financial reward to the investors willing to take the risks associated with its development.

REFERENCES

1. WILLIAMS, P.M., et al., "MHTGR Development in the United States", Progress in Nuclear Energy, ISSN 0149-1970, Vol. 28, No. 3, (1994), pp.265-346.

2. BREY, H.L., (Ed), "Current Status and Future Development of Modular HTGR Technology", IAEA-TECDOC-TBD, Vienna, 2000.

3. INTERNATIONAL ATOMIC ENERGY AGENCY, Gas Cooled Reactor Design and Safety, Technical Report Series No. 312, IAEA, Vienna (1990).

4. RENNIE, C.A., Achievements of the Dragon Project, Ann. Nuclear Energy 5 (8) (1978), pp.305-320.

5. MOORE, R.A., et al., "HTGR Experience, Programs, and Future Applications", Nuclear Engineering and Design 72 (1982), pp. 153-174.

6. KROGER, W., et al., "Safety Characteristics of Modern High-Temperature Reactors: Focus on German Designs", Nuclear Safety, Vol. 29, No. 1, (1988) pp. 36-48.

7. SILADY, F.A., PARME, L.L., "The Safety Approach of the Modular High Temperature Gas-Cooled reactor (MHTGR)", 11th International Conference on the HTGR, Dimitrovgrad, USSR, (13-16 June, 1989).

8. SILADY, F.A., GOTSCHALL, H., "The 450 MWt MHTGR: More Power with the Same High Level of Safety", Third International Seminar on Small and Medium Sized Nuclear Reactors, New Dehli, India (26-28 August, 1991)

9. BRANDES, S., KOHL, W., "HTR 100 Industrial, Nuclear Power Plant for Generation of Heat and Electricity", IAEA-TECDOC-436, Vienna, 213-234, (1987)

10. INTERNATIONAL ATOMIC ENERGY AGENCY, "Design and Development of GCRs with Closed Cycle Gas Turbines", Vienna, IAEA-TECDOC-899, August 1996.

11. INTERNATIONAL ATOMIC ENERGY AGENCY, "High Temperature Gas Cooled Reactor Technology Development", IAEA-TECDOC-988, Vienna, December 1997.

12. LA BAR, M.P., SIMON, W.A., "International Cooperation in Developing the GT-MHR", IAEA-TECDOC-988, Vienna, December 1997, 59-71.

13. BREY, H.L. (Ed), "Helium Gas Turbine Reactor Technical Challenges; A Characterization for Use in Focusing R&D Resources", EPRI document TP-114690, Palo Alto, California, January 2000

14. SIMON, W.A., SHENOY, A.S., "International Cooperation in Developing the GT-MHR, Evolution and Program Status", proceedings on the IAEA Technical Committee Meeting on High Temperature Gas Cooled Reactor Applications and Future Prospects, ECN-R-98-004, Petten, the Netherlands, 10-12 November 1997.

15. HAVERKATE, B.R.W, VAN HEEK, A.I., KIKSTRA, J.F., "An HTR Cogeneration System for Industrial Applications", proceedings of a Technical Committee Meeting, Beijing, China, 2-4 November 1998, IAEA-TECDOC (to be published)

16. MUTO, Y, "Present Activity of the Design and Experimental Works for HTGR-GT System in JAERI", EPRI Helium Gas Turbine Reactor Workshop, Palo Alto, California, December 7-8, 1999

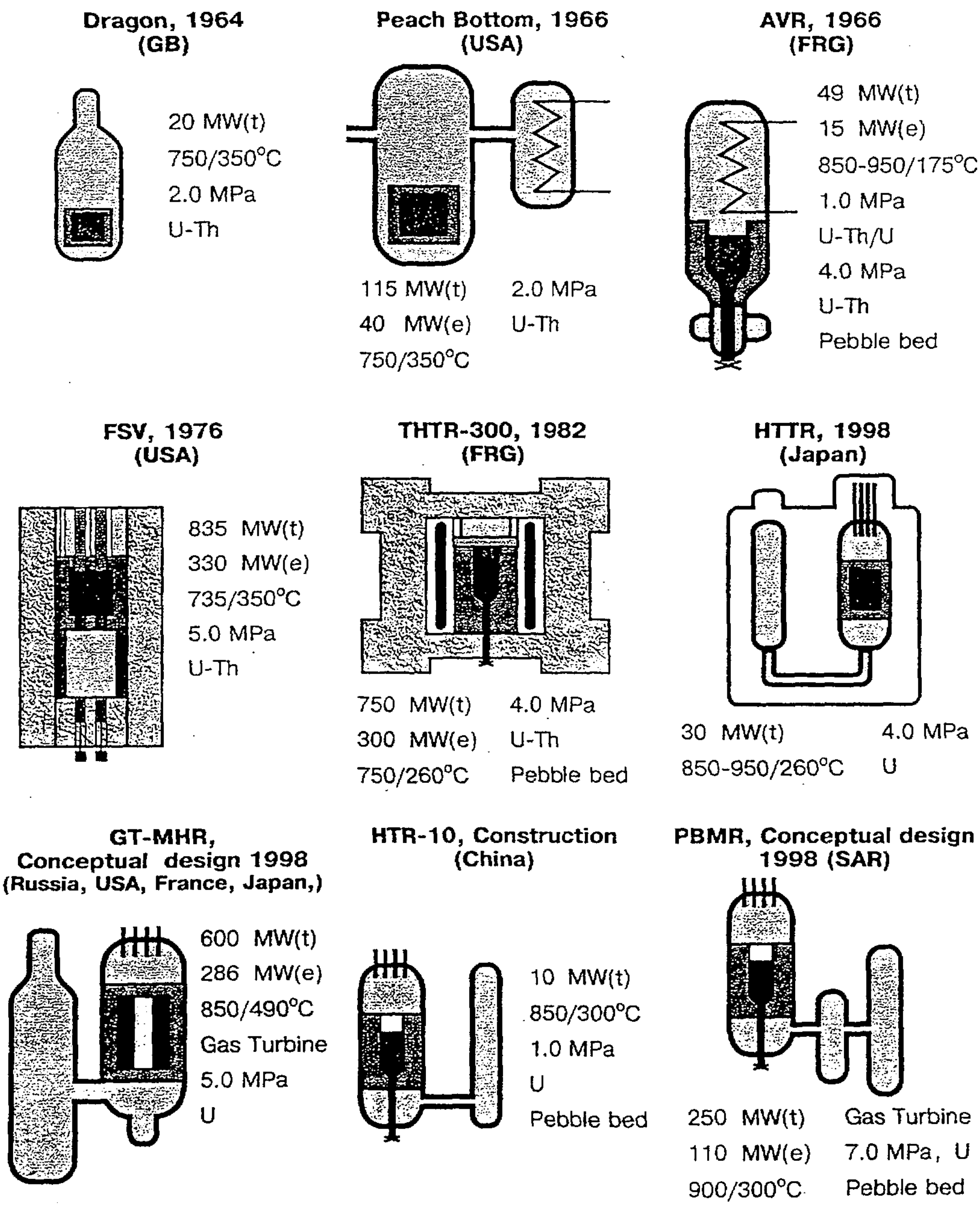
17. YAN, X., SHIOZAWA, S., KUNITOMI, K., MUTO, Y., MIYAMOTO, Y., "Design of Gas Turbine High Temperature Reactor-GTHTR 300", EPRI Helium Gas Turbine Reactor Workshop, Palo Alto, California, December 7-8, 1999

18. MASSACHUSETTS INSTITUTE OF TECHNOLOGY and IDAHO NATIONAL ENGINEERING & ENVIRONMENTAL LABORATORY, "Modular Pebble Bed Reactor Project", First Annual Report, (June 1999).

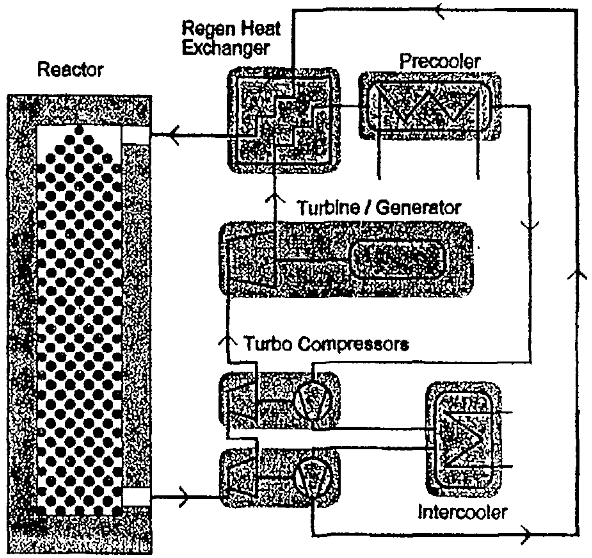
19. CLEVELAND, J., LEWKOWICZ, I., "Status of the IAEA Coordinated Research Programme on Design and Evaluation of Heat Utilization Systems for the HTTR", Presented at the 2nd International Conference on Multiphase Flow, Kyoto, Japan, (April 1995)

20. BAUMER, R., KALINOWSKI, I., "THTR Commissioning and Operating Experience", Energy, Vol. 16, No.1/2, (1991), pp.59-70

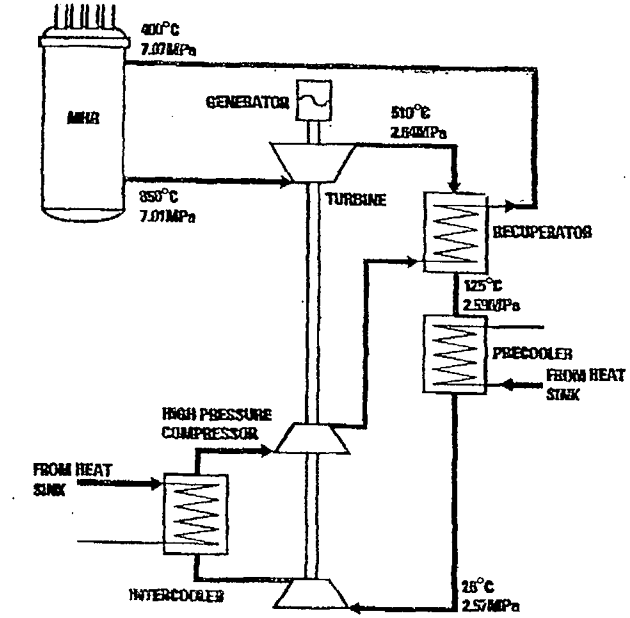
He Cooled HTGR in World
ВТГР с гелиевым охлаждением в мире



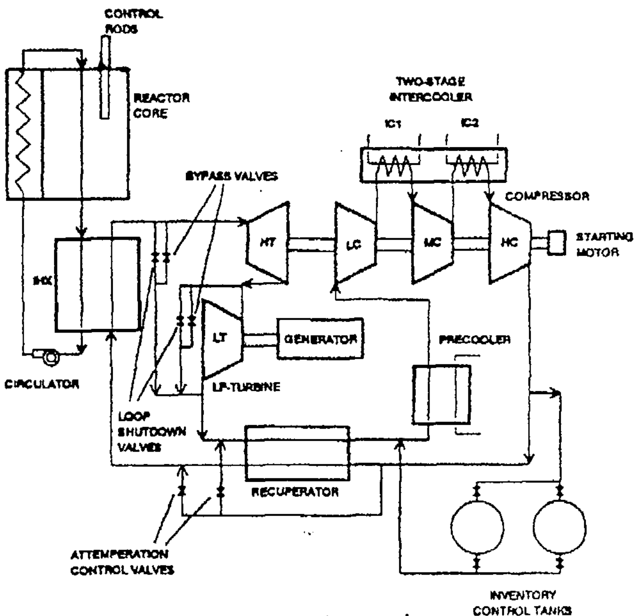
HTGRs in the world
Courtesy of:
V. Grebennik, Kurchatov Institute



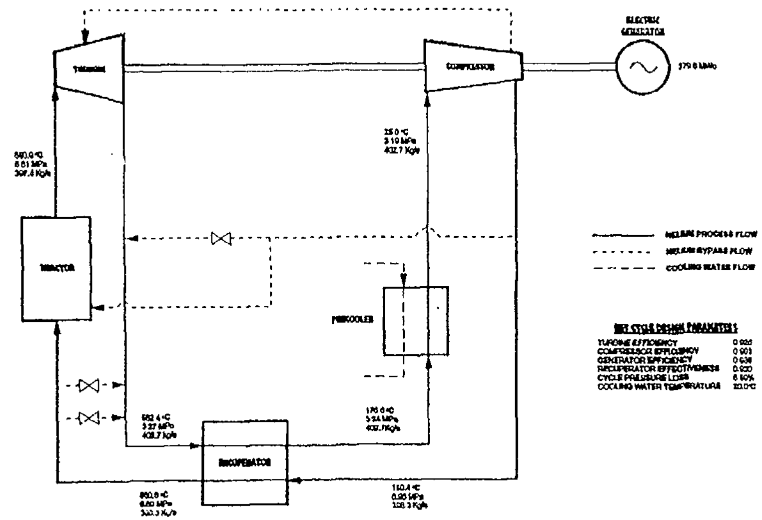
PBMR System Arrangement



GT-MHR System Arrangement

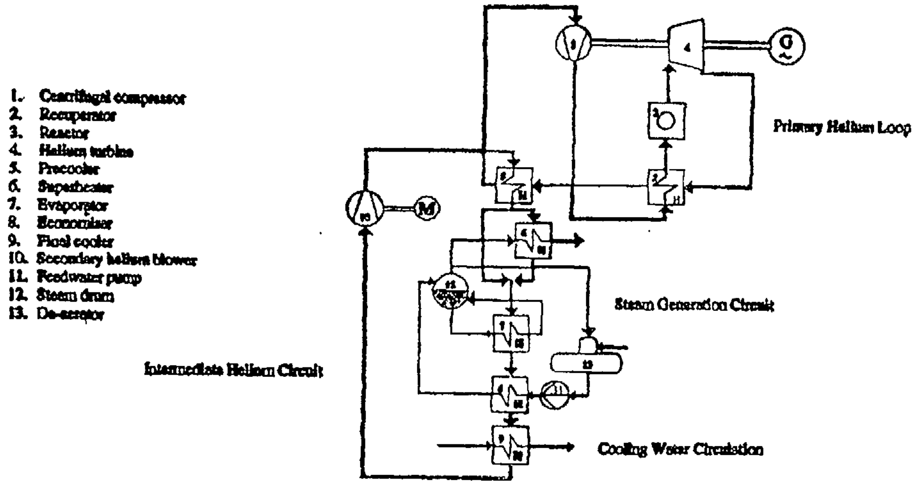


MPBR Flow Schematic

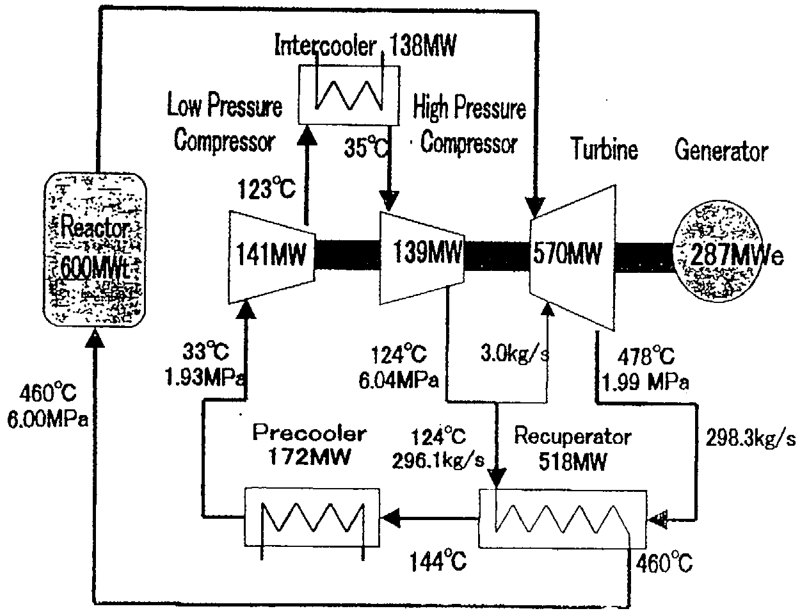


System arrangement of GTHTR-300

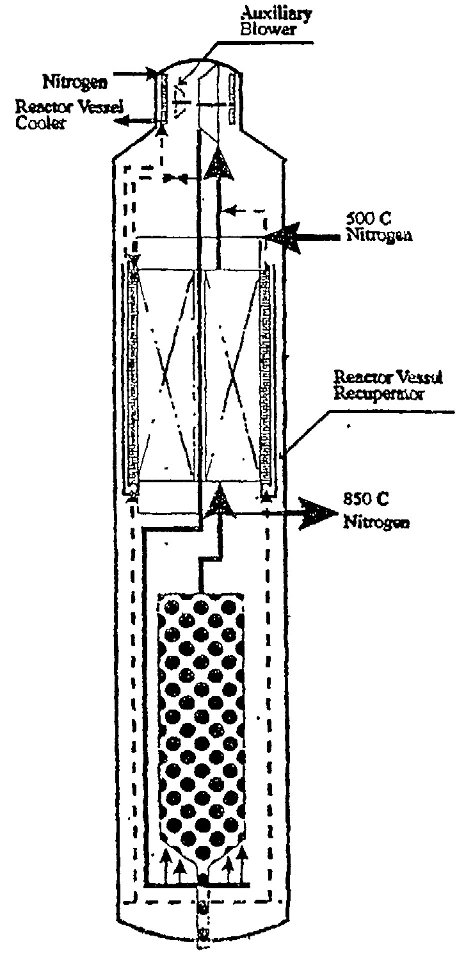
System Arrangements for Selected Modular HTGR Gas Turbine Plant Designs



Flow Diagram of ACACIA Co-generation Plant



System Arrangement of 600 MW HTGR-GT



Primary System Layout
of
MHTGR-IGT

System Arrangements
For
Selected Modular HTGR Gas Turbine Plant Designs

Appendix B

Characteristics of Selected Steam Cycle HTGRs

	AVR	Peach Bottom	Ft. St. Vrain	THTR-300	HTR-500	VGM-400	HTR-Module	MHTGR
Country of Origin	Germany	U.S.	U.S.	Germany	Germany	Russia	Germany	U.S.
Thermal Power, MWt	46	115	842	750	1,390	1,060	200	350
Net Elec. Power, MWe	13	40	330	300	550	Co-Gen.	80	139
Power Density, MW/m^3	2.5	8.3	6.3	6.0	6.6	6.9	3.0	5.9
Core Outlet Temperature, °C	950	725	775	750	700	950	700	686
Helium Pressure, MPa	1.1	2.25	4.8	3.9	5.5	5.0	6.6	6.4
Steam Temp. °C	505	538	538/538	530/530	530	535	530	538
Elec. Gen., MWh	~ 1,670	~ 1,380	~ 5,500	~ 2,890	N/A	N/A	N/A	N/A
Reactor Type	Pebble	Sleeve	Block	Pebble	Pebble	Pebble	Pebble	Block
Fuel Enrichment	Various	HEU	HEU	HEU	LEU	LEU	LEU	LEU
Fuel Composition	Oxide	Carbide	Carbide	Oxide	Oxide	Oxide	Oxide	Ox-Carb
Fuel Coating	Various	BISO	TRISO	BISO	TRISO	TRISO	TRISO	TRISO
Vessel Material	Steel	Steel	PCRV	PCRV	PCRV	PCRV	Steel	Steel

Characteristics of Selected Modular HTGR Gas Turbine Plants

	GT-MHR	PBMR	MHTGR -IGT	ACACIA	GTHTR-300	600MW-HTGR-GT	MPBR
Country of Origin	U.S./ Russia	S. Africa	China	Netherlands	Japan	Japan	U.S.
Thermal Power, MWt	600	265	200	40	600	600	250
Net Elec. Power, MWe	278	116	~ 96	Co-Gen.	273	287	112
Pwr. Density, MW/m^3	6.5	4.3	3.0	______	______	5.77	______
Core Outlet Temp., °C	850	900	900	800	850	850	850
Helium Pressure, MPa	7.15	7.0	6.0	2.3	6.8	6.0	7.9
Cycle Type	Direct	Direct	In-Direct	Direct	Direct	Direct	In-Direct
Core Type	Block	Pebble	Pebble	Pebble	Block	Pin/Block	Pebble
Fuel Enrichment	HE-Pu	LEU	LEU	LEU	LEU	LEU	LEU
Fuel Composition	PuO	Oxide	Oxide	Oxide	Oxide	Oxide	Oxide
Fuel Coating	TRISO	TRISO	TRISO	TRISO	TRISO	TRISO	TRISO
Vessel Material	Steel	Steel	Steel	Steel	Steel	Steel	Steel